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THE ACCIDENT AT CHERNOBYL AND ITS IMPLICATIONS FOR THE SAFETY OF CANDU REACTORS

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Canada

Report

FOREWORD

The Atomic Energy Control Board (AECB) is the agency of the Government of Canada which controls the development, application, and use of atomic energy, and participates on behalf of Canada in international measures of control. The AECB exercises the authority vested in it by the Atomic Energy Control Act by means of general regulations, regulatory policy statements and guides, and licences for specific facilities. In the area of nuclear power reactors, the owner/operator of any nuclear reactor in Canada must obtain a construction approval and, prior to operating the reactor, an operating licence. The AECB maintains a staff in Ottawa and at each major nuclear reactor site in Canada to review licence applications and to monitor compliance with the conditions of the licences.

Immediately after the serious accident at the Chernobyl nuclear reactor complex in the Ukrainian Soviet Socialist Republic, the AECB attempted to obtain as much information as was available, and to review the implications of the accident. Regulatory agencies and organizations involved in nuclear power in most countries were pursuing the same objective. However, at that time, little technical information was available on the causes of the accident.

On May 15 1986, shortly after the accident, a debate on nuclear power was held in the Canadian House of Commons. In the course of that debate, the Minister of Energy, Mines, and Resources stated that she was asking the AECB, and other responsible Canadian agencies, to participate in the international assessments of the Chernobyl accident, and to report fully on its implications. In formally asking the AECB to do this, the Minister requested that the AECB ascertain what lessons can be learned with respect to the safety of CANDU nuclear reactors and the regulatory process in Canada, and to report on the implications of the Chernobyl incident for Canada.

A post-accident review meeting of nuclear power and radiation protection specialists was convened by the International Atomic Energy Agency in Vienna during the week of August 25-29, 1986. At the meeting, representatives from the USSR tabled reports and described and responded to questions on the accident, its causes and consequences. Two members of the staff of the AECB were included in the Canadian delegation to the meeting.

On the basis of the information presented at that meeting, the AECB carried out a study of the design of the Chernobyl reactor, the events which led to the accident, and its consequences. The causes of the accident were carefully examined to ascertain whether they revealed any shortcomings in the safety of CANDU reactors and the Canadian regulatory process. The study and its conclusions are documented in this report, which has been prepared by the staff of the AECB in consultation with other responsible Canadian agencies to fulfil the Minister's request.

EXECUTIVE SUMMARY

Introduction

On 26 April, 1986, a serious accident occurred at a nuclear power reactor at Chernobyl in the Ukrainian Soviet Socialist Republic. The accident caused the death of 31 plant workers and firemen, and the release of large quantities of radioactive materials which spread over the surrounding area and, in lesser amounts, to other parts of the USSR and to other countries.

In August 1986, a delegation of Canadians, including two members of the staff of the AECB, attended a post-accident review meeting in Vienna, at which Soviet representatives described the accident and its causes and consequences. On the basis of the information presented at that meeting, AECB staff conducted a study of the accident to ascertain its implications for the safety of CANDU nuclear reactors and for the regulatory process in Canada.

The Chernobyl Reactor

The Chernobyl reactor was of a type referred to as RBMK in which the neutron-moderating material is graphite and the fuel is cooled by ordinary water (in CANDU reactors, heavy water is used for both these functions). In the RBMK design, steam is produced directly in the fuel cooling system, whereas the CANDU utilizes steam generators in which the heat removed from the fuel by the heavy water is used to boil ordinary water.

In the Chernobyl reactors, much of the fuel cooling equipment is contained within leak-tight compartments which are intended to fulfil a function similar to that of the concrete containment buildings in which CANDU reactors are required to be housed. However, the reactor and the piping which exits above the top of the reactor is not contained in such a compartment.

The RBMK design includes emergency protective mechanisms which are designed to shut the reactor down if a hazardous condition is detected. The rate of shutdown is rather slow, and the success of the protective action depends on supplementary action by control rods which normally are partially inserted into the reactor.

A feature of the reactor which played an important part in the accident was a characteristic known in technical terms as a positive void coefficient of reactivity. This means that when the amount of steam (or void) in the reactor increases, there is a feedback effect which, if not controlled, will cause the power of the reactor to increase, and thereby further increase the void in the reactor.

The Accident

In the course of shutting down the reactor for routine maintenance, the operators were conducting a test to demonstrate that, after the steam supply to a turbine had been cut off, the mechanical energy of its rotor could be harnessed to supply electrical power to important equipment for about 50 seconds.

While reducing power in preparation for this test, the operators made a simple error which caused the reactor power to be reduced to a low level, at which very little steam was being produced in the reactor. Under the conditions which existed at that point, the characteristics of the reactor made it difficult to re-establish the required power, and the test should have been aborted.

In an effort to increase reactor power to a level at which the test could be performed, the operators committed several serious violations of the rules for safe operation of the reactor. In particular, they blocked several of the signals which could have initiated an emergency shutdown, and removed from the reactor most of the control rods which are required to supplement the emergency shutdown action. These violations seriously impaired the capability of the emergency shutdown equipment.

The operators succeeded in restoring power to a level at which they believed they could conduct the test, although the condition of the reactor was still such that there was very little steam in the core. When the test was initiated, the flow of coolant in the reactor decreased and the rate of steam production began to increase. Because of the positive void coefficient of reactivity discussed above, the steam production produced a feedback effect which caused the reactor power to increase. The control system was unable to cope with this power increase, and under these conditions the emergency shutdown equipment could not act quickly enough to terminate it. The result was a rapid uncontrolled rise in reactor power which was terminated only when there was a violent explosion which destroyed the reactor.

The building above the reactor was destroyed, and large quantities of radioactive materials were released. Many fires were started, and these were extinguished only at a severe cost to the firefighters in terms of exposure to radiation. The graphite moderator ignited, and it took about 10 days before the Soviets were successful in controlling the situation and terminating the major releases of radioactive substances.

Radiological Releases and Health Effects

In addition to two plant workers who were killed about the time of the accident about 200 workers and firemen suffered acute radiation syndrome, and of these, 29 died. The radiation levels in the vicinity were sufficiently high that an area of 30 km radius was evacuated. Radioactive materials in lesser quantities spread to other parts of the USSR and to other countries.

Implications for Safety and Licensing of Canadian Reactors

The staff of the AECB has carefully studied the Chernobyl accident to ascertain whether there are lessons to be learned relevant to the safety or the licensing requirements of reactors in Canada. The objectives of this review have been first to examine critically the similarities between the RBMK reactors and CANDU reactors in order to determine whether this accident has revealed any weaknesses in the safety or licensing requirements of Canadian reactors, and second to examine the operating errors to determine whether similar errors could be made by Canadian operators. The study has addressed the design of the reactors and their safety systems, the operation of the reactors including the processes for preventing violations of operating rules, and other aspects such as emergency preparedness.

The conclusion of this review is that the accident at Chernobyl has not revealed any important new information which would have an effect on the safety requirements for CANDU reactors as presently applied by the AECB. All important aspects of the accident and its causes have been considered by the AECB in the licensing process for currently licensed reactors. However, a number of recommendations are made with respect to aspects of reactor safety which should be re-examined in order to reinforce this conclusion. These recommendations are listed in section 9 of this report.

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THE ACCIDENT AT CHERNOBYL AND ITS IMPLICATIONS FOR THE SAFETY OF CANDU REACTORS

1. Introduction

On April 26, 1986, a serious accident occurred at the fourth unit of the nuclear power reactor complex at Chernobyl, in the Ukrainian Soviet Socialist Republic. The accident caused the death of 31 plant workers and firemen and the release of large quantities of radioactive materials which spread over the surrounding area and, to a lesser extent, to other parts of the USSR and to other countries.

A meeting of nuclear power and radiation protection specialists was convened by the International Atomic Energy Agency (IAEA) in Vienna during the week of August 25-29, 1986. At this meeting, representatives from the USSR tabled reports and described and responded to questions on the accident, its causes and consequences. The Canadian delegation included two representatives of the Atomic Energy Control Board.

This report summarizes the design of the Chernobyl reactor, the events leading up to the accident, and its consequences. It compares design features of the Chernobyl reactors with those of CANDU reactors. Finally, it discusses implications of the Chernobyl accident for the safety and regulation of CANDU reactors in Canada, from the perspective of the Atomic Energy Control Board.

This report is not intended to be a comprehensive description of the Chernobyl reactor or of the accident that occurred on April 26. Many other documents (for example references 1 and 2) are available and contain detailed information on these topics. The purpose of this report is rather to summarize the important features of the Chernobyl design and the major causes of the accident in order to assess any major implications for the licensing and operation of reactors in Canada.

The AECB study of the Chernobyl accident has been based largely on the information presented by the Soviet representatives at the post-accident review meeting. This information appears to be quite comprehensive in the technical description of the reactor design, the sequence of events which led to the accident, and the remedial actions and consequences. However, some aspects were not described in detail and therefore cannot be addressed in this report. These include the details of the reactor's control panels, the Soviet approach to operator training, and the responsibilities and lines of authority for operation of the reactor.

The accident at Chernobyl had important political ramifications, and had implications on public opinion on the safety of nuclear power in many countries. As a result of the accident, a special program on nuclear safety was initiated by the IAEA, and two international conventions on early warning and mutual assistance in the case of a nuclear accident were drafted. Furthermore, the accident caused considerable disruption in international trade in agricultural commodities. This report addresses only the technical aspects of the accident which are relevant to the safety and regulation of nuclear power reactors. It does not address the broader political issues or the effect of the accident on public opinion.

The discussion of the implications for reactor licensing in Canada is limited to the implications for commercial CANDU power reactors, and does not address any considerations related to research reactors or to the relatively low powered Nuclear Power Demonstration reactor (NPD) at Rolphton, Ontario. A separate report addressing the implications for the NPD reactor will be prepared by the AECB.

2. Reactor Design Concepts

In order to explain the events which caused the Chernobyl accident, it is necessary to introduce the following fundamental aspects of the design of nuclear reactors.

2.1 Nuclear Fission

The basic function of a power reactor is to provide the conditions necessary for splitting atoms of uranium and for utilizing the large quantities of heat produced in this way to generate electricity. The splitting or fission process as it is termed occurs when a neutron travelling at a relatively low speed collides with a uranium nucleus. Immediately following a fission, some (usually 2 or 3) additional fast neutrons are released, which may, after being slowed down by repeated collisions with a suitable material, collide with another uranium nucleus to cause another fission. By controlling the extent to which this chain reaction is able to multiply successive generations of neutrons it is possible to control the reactor at a constant power level.

2.2 Reactivity

An important concept which is used to express the effectiveness of control rods or other absorption mechanisms is their reactivity worth.

Reactivity is a measure of the rate at which a reactor will increase or decrease the number of fissions and consequently the power generated in the fuel. The reactivity of a reactor varies mainly as a result of the amount of absorption that neutrons encounter as they diffuse through the moderator before colliding with another uranium atom. A reactor is said to be "critical" when this absorption is just sufficient to remove the extra neutrons created during a fission, leaving one neutron to sustain the chain reaction.

The unit of reactivity used in Canada and subsequently in this report is the milli-k which is further abbreviated to "mk".

2.3 Void Reactivity

An important factor which had a significant impact on the accident is the effect of the steam content in the coolant channels on the absorption of neutrons within the core. As the coolant is heated by an increase in power, it forms steam, which causes its density to decrease. The resulting reduction in neutron absorption increases reactivity, which further increases power and reduces density. This process is referred to as the positive void reactivity effect, and it has the potential, if not controlled, to cause a very rapid increase in reactor power.

3. Design of the RBMK Reactor

3.1 General Plant Description

The Chernobyl nuclear power complex comprises four nuclear reactors, eight turbogenerators and associated equipment, services and facilities arranged within a single long building. The design power output of each reactor is 1000 MW(e). The total thermal power of each reactor is 3200 MW(t).

3.2 Reactor

The reactor is of a type referred to by the Soviets as RBMK, which means a large channel-type power reactor. A simplified diagram of it is shown in Figure 1.

The reactor core is a vertical cylinder approximately 11.8 m in diameter and 7 m high. It is composed of 1661 vertical pressure tubes with the fuel assemblies inside them, a graphite moderator, 211 channels with neutron absorber rods (control and protection system), and the sensors of the control system. The graphite moderator consists of blocks assembled into columns, with vertical holes through them into which the pressure tubes and control and protection system channels are inserted.

The metal structure at the top of the reactor is covered by a removable floor consisting of individual slabs of concrete. This allows the fuelling machines to gain access to the fuel channels to carry out on-power refuelling. The concrete is a radiation shield for the area above the reactor.

3.3 Heat Transport Circuit

The heat generated by the reactor fuel produces steam by causing water to boil in the heat transport circuit. This steam is subsequently separated from the water, and the heat stored in it is converted to electrical energy in the turbogenerators. A simplified schematic of the heat transport circuit is shown in figure 2. The circuit consists of two loops, similar in their arrangement and equipment, which function in parallel and independently to remove heat from half of the reactor's fuel assemblies. Each loop includes two steam separators, four main circulation pumps, and the associated piping and valves to direct the cooling water to and from the fuel channels.

Under normal operating conditions at full power, three of the four main circulating pumps in each loop are in operation, with one held in reserve. Water is pumped by the main circulating pumps through piping, which directs it to headers, and hence into the fuel channel inlets. As it passes through the fuel channels, the water surrounding the fuel elements is heated to saturation temperature, partially evaporates (14.5% on average) and the steam-water mixture flows through the individual steam lines into the separators, where it is separated into steam and water. The steam passes to the turbines while the water is mixed with feedwater (condensed steam from turbogenerator), and returns to the main circulating pumps.

3.4 Containment

The containment or confinement system of the fourth unit is designed to confine radioactive releases during accidents involving failure of any piping of the reactor cooling circuit, with the exception of the steam-water lines, upper ends of the fuel channels, and parts of the downcomers, which are all outside of the containment. Basically it consists of leak tight compartments, surrounding some components of the heat transport circuit, and a condensation-type pressure suppression system.

The system is designed to cope with the steam-water discharge associated with a break in the largest size of piping (i.e., headers), but is apparently unable to cope with failure of a large number of pressure tubes.

The areas above the reactor vault, which house the steam generators, upper ends of the fuel channels, feeders, top portion of the downcomers and other interconnecting piping, are outside of the confinement and therefore have essentially no heat suppressing or pressure retaining capability.

3.5 Reactor Emergency Cooling System

The emergency core cooling system (ECCS) is a safety system designed to cool the fuel (after shutdown of the reactor) by feeding water into the reactor channels in the event of accidents in which there is a rupture of the core cooling system.

The ECCS system comprises three independent subsystems, each of which ensures not less than 50% of the required output. Each subsystem includes a fast-acting part for the initial stage of the accident and a part providing prolonged heat removal capability.

The fast-acting parts of two ECCS subsystems take the form of a system of vessels filled with water or nitrogen at high pressure connected by pipes to the primary coolant system. The fast-acting part of the third ECCS subsystem is a unit for supplying water from an electric feed pump. A separate pumping system supplies long-term cooling to the reactor core.

Standby power for the electric motors of the pumps and fast-acting valves of the ECCS is supplied by the diesel generator. Because the electric feed pump must have an uninterrupted power supply, it was designed to operate for the first 45-50 s on rundown power from the turbogenerator. This was the mode of operation to be verified in the test being performed at the time of the accident.

3.6 Reactor Physics and Protective/Control Systems

3.6.1 Moderator and Fuel

In the RBMK reactor, the moderating material that slows down the neutrons is composed of graphite, and the fuel that undergoes fission is uranium oxide enriched in the naturally occurring isotope uranium 235. The fuel, arranged in long bundles of elements located inside pressure tubes, is cooled by the flowing mixture of high-pressure steam and water.

3.6.2 Protective and Control Systems

To control the reactor, neutron absorbers in the form of long cylindrical rods are inserted or withdrawn from the reactor core. The several classes of these rods and their function in the RBMK reactors are described as follows:

a) Emergency Rods - a group of 24 control rods which are poised above the reactor and are automatically driven into the core in the event of a hazardous condition being detected.

b) Manual Rods - a group of 139 manually operated rods which are used principally to control the shape of the neutron distribution and therefore the power distribution through the core. They are also driven into the core in the event of a hazardous condition being detected.

An auxiliary function of these rods is to compensate for the increased or decreased absorption of neutrons which occurs for several hours following a change in reactor power level. This change in absorption derives from the variation in the amount of xenon 135, a product of the fission process, which is a strong absorber of neutrons, and which is consequently referred to as a "poison".

c) Regulating Rods - There are two groups of 12 automatic regulating rods in the RBMK reactor.

In the first of these groups, the Local Automatic Rods (LAR), individual rods are located symmetrically in the reactor core. By automatically inserting or withdrawing these single rods, the tendency for the reactor power distribution to vary among these regions can be controlled.

At relatively low power levels the RBMK reactors can be controlled automatically using the second group of rods, the Automatic Regulating (AR), which are divided into 3 separate banks to control the bulk power level.

d) Axial Shaping Rods - a group of 24 rods which are inserted into the core from the bottom and shape the power distribution along the length of a fuel channel.

While the 24 control rods selected as the emergency rods are reserved for insertion into the reactor in the event of an accident, the manual rods and regulating rods are also automatically driven into the core when a hazardous condition is detected.

4. The Accident

The accident at Chernobyl occurred when the reactor power increased in an uncontrolled fashion during a test being carried out on electrical equipment.

4.1 Purpose of Test

The purpose of the test was to demonstrate that the mechanical energy of a turbogenerator cut off from its steam supply could be used to sustain the reactor's own electrical power requirements when normal power supplies are not available. This was apparently a commissioning test that had not been completed successfully during the initial commissioning of the reactor.

Specifically, the test was intended to demonstrate that if a turbogenerator had its steam supply cut off, it could continue to supply electrical power to run the emergency pumps during the 50 seconds required to start up the standby diesel generators. This was the second such test, the first one (on unit 3) having been unsuccessful, about one year previously, when it was found that the output voltage dropped too quickly, before the mechanical energy of the rotor was expended. To correct this problem, a modification had been made to the turbogenerator's electrical controls.

4.2 Test Procedure

The test called for the reactor power to be reduced to about 700 MW(t) with all the steam being diverted to one turbogenerator (Number 8), with the other one shut down. Number 8 turbogenerator was then to be isolated by closing its steam stop valves, and electrical power to important equipment was to be supplied from the turbogenerator by utilizing the mechanical energy of its rotor. After the test, a planned maintenance outage was to start.

The procedures for this test were not properly prepared, and were not reviewed or approved by those with the required authority and expertise in reactor safety. Furthermore, as noted below, the operators made several important deviations from the test procedure, and from the rules and practices for safe operation of the reactor.

While the test was intended to demonstrate that an important aspect of the design would work in practice, the operators had been instructed that if the test was not completed successfully prior to this maintenance shutdown, they would not be permitted to repeat it until the following maintenance shutdown about one year later. The operators were therefore under considerable pressure to complete the test successfully before the plant was shut down.

4.3. Events Leading to the Accident

In this section, the salient aspects which caused the accident at the fourth unit of the Chernobyl Nuclear Power Plant are summarized. A detailed accident sequence is presented in Appendix "A".

On April 25, 1986, at 01:00, the staff began to reduce the reactor power in preparation for the test. At 13:05, turbogenerator Number 7 was switched off with the reactor at 1600 MW(t). The electric power supply for the unit's own needs was switched to turbogenerator Number 8.

At 14:00, the reactor's emergency core cooling system was disabled in accordance with the test procedure. This system remained disconnected through the remainder of the test, although its unavailability played no significant role in the incident. Because of the off-site power demand, the operators were requested to postpone the test and to maintain power at 50% full power. While power was held at this level, the amount of xenon poison in the fuel increased, necessitating the withdrawal of some control rods.

At 23:10, the power decrease was continued towards the target level of between 700 MW(t) and 1000 MW(t). The operators switched control from a system which automatically controlled the distribution of power within the reactor to a system used at lower power, which controlled the total power without compensation for local effects. However, in doing so the operator failed to adjust a setpoint, with the result that the control system reduced reactor power rapidly to 30 MW(t). This was in fact the only unintentional operating error, as other deviations from the test procedure and from safe operating practices were intentional violations committed with the objective of successfully completing the test.

This sharp reduction in power caused most of the steam in the channels to condense, so that most of the void in the fuel channels collapsed. Furthermore, it resulted in an increase in the xenon poisoning effect. Both of these effects caused an increase in the absorption of neutrons within the reactor and so reduced the reactivity. The only recourse for the operators to keep the reactor operating to conduct the test was to withdraw manual control rods from the reactor. By doing so they were able to increase the reactor power, but only as high as 200 MW(t).

According to the station operating procedures, the operators should have aborted the test and shut down the reactor, since this withdrawal of the manual control rods had reduced the operating reactivity margin well below the permitted limit of 30 manual rods (or their equivalent) inserted. Furthermore, the reactor power was at a low level (less than 700 MW(t)) at which continued operation was not permitted because at such power levels the reactor was particularly unstable.

Nevertheless, it was decided to proceed with the test. At 01:03 (April 26) and at 01:07, the two additional main circulation pumps were switched on to join the six pumps already operating. The resulting increase in flow through the core caused a reduction in steam formation, a fall in steam pressure, and unstable water levels in the steam separators. Operation at these higher flows was another violation of operating policy. The operators attempted manually to sustain the level in the separators by increasing feedwater flow to three times the normal value, but were only moderately successful. The operators blocked the emergency protection signals relating to low steam separator level and low steam separator pressure, again to avoid the possibility of shutting down the reactor before the test was completed. This was also a violation of operating policy.

The reactivity continued to drop due to a decrease in the inlet temperature and decreased void in the channels. To combat this, the automatic control rods withdrew further from the reactor and approached their upper limits. The operators withdrew manual rods to cause the automatic rods to be re-inserted into the core.

Since the pressure in the steam separator continued to decrease, the operator closed the steam bypass line, which can be used to send excess steam production directly to the condenser or to redirect the steam completely in case of a turbogenerator trip. As the level of water in the steam separator continued to rise, the operator abruptly decreased the flow from the feedwater circuit at approximately 01:22. This resulted in an increase in steam formation in the fuel channels as more warm water from the separator reached the core. To compensate for the increased reactivity due to increased void in the core, the automatic control rods started driving farther. For the next minute the reactor appeared to stabilize, the automatic control rods stopped moving, the rate of steam formation levelled off, and the steam separator levels were stable.

At this stage, a computer printout showing the status of important reactor parameters indicated clearly that the reactivity margin was at a level at which the operating procedures required the reactor to be shut down immediately. The operators violated this requirement in order to carry out the test before the reactor was shut down.

Prior to commencing the test, the operators committed one further violation of procedure by blocking the reactor shutdown which would occur when both turbine generators were tripped. This was done so that the reactor would continue to operate and thus allow the test to be repeated if necessary.

The reactor was at this point in a very unusual condition, in which control was difficult and the reactor was potentially very unstable. The fuel channels were essentially full of water at or near its boiling temperature, and the eight circulating pumps were maintaining an unusually high rate of flow through them. The reactor power was relatively low (about 7% of full power), and the distribution of the neutron flux (and power) was unusual in that it was depressed at the centre and peaked towards both the top and bottom. The automatic control rods were partially inserted; however, their remaining reactivity worth was relatively low, particularly in comparison to the potential positive reactivity effect which could result from an increase in void. The emergency shutdown actions on abnormal steam separator level and pressure, as well as that on loss of both turbogenerators, had been disabled, so that the only available emergency shutdown signals were those on reactor power level and rate of change of power level. Furthermore, the manual control rods had been withdrawn too far, so that the reactivity margin was too low and the rate at which the shutdown system could act in the event of an accident was greatly reduced. Finally, the configuration of the reactor had increased the void reactivity effect by about 50%.

The test commenced at 01:23 when the operators closed the valves in the steam lines to the turbogenerator. This caused a gradual reduction in flow to the reactor as those pumps whose power was supplied by this turbogenerator began to run down. The reactor remained at power, because the shutdown signal on loss of the turbine had been disabled.

The immediate consequence of closing the valves was an increase in the pressure in the heat transport system, which resulted in a further reduction of void in the channels. One of the banks of automatic regulating rods began to withdraw to compensate for this small decrease in reactivity. However, the reduction in cooling flow through the reactor caused by the pumps running down

subsequently caused an increase in void and a consequent increase in reactivity. This effect was sufficiently large that all three banks of automatic control rods began to drive into the reactor to compensate. At about 01:23:40, the automatic control rods could no longer compensate for the reactivity increase caused by the increase in void, so that the reactivity began to increase and reactor power began to rise.

At about this time, the operator attempted to trip the reactor manually by pressing the emergency shutdown button, which would cause all the control and emergency rods to drive into the reactor. However, since the operating reactivity margin had been greatly reduced, and since the rate of insertion of the rods was relatively slow, this action was not effective in preventing the subsequent power surge.

As the power increased, the rate of steam formation was rising, causing a further increase in the void in the reactor. Because of the positive void-reactivity feedback effect, the increase in void itself was causing the power to increase still more. These two effects combined to produce a positive feedback which caused the reactor power to increase very rapidly.

It was later estimated by the Soviets, on the basis of reactor physics calculations, that under these conditions the reactor power would increase to more than 100 times full power within a few seconds. Such a large surge in power would cause the fuel to fragment violently and produce large quantities of steam, and the resulting very high pressures would destroy the reactor. The actual power level attained, and the point at which the explosion occurred, are a matter of speculation. It is also possible that, because of the large size of the reactor and the distorted power distribution, the initial power surge occurred only in one part of the reactor. In any case, the result was an extremely violent steam explosion, followed a few seconds later by a second explosion, which is believed to have been caused by ignition of hydrogen produced by chemical reactions at the very high temperatures. The reactor was destroyed by these explosions.

Witnesses outside the building heard the two successive explosions, and saw burning material and sparks shooting into the air above the reactor. Some of this material fell on the roof of the turbine building and caused approximately thirty fires.

4.4 Subsequent Events

The most urgent problem immediately after the explosion was the fires which threatened to spread to other parts of the station. Heroic efforts by the fire-fighters were successful in containing these fires and extinguishing them by about 05:00. However, these fire-fighting activities were carried out in areas where there were high radiation fields and severe radioactive contamination. As a result, many fire-fighters received very large doses of radiation, some of which subsequently proved fatal.

The explosions destroyed the top of the reactor, lifting off the concrete shield and knocking the refuelling machine on its side. A videotape taken by the Soviets shows very extensive damage to the reactor and associated equipment.

The damage caused by the explosion allowed air to enter the reactor, causing the hot graphite to burn. This increased the release of fission products from the hot fuel. In an attempt to extinguish this fire and reduce the release of radioactive material, the Soviets dropped large quantities of material from helicopters on the reactor. The materials dropped were boron (a neutron absorber, to ensure that the reactor remained shut down), dolomite, lead, sand, and clay. While this was initially successful, it also reduced heat losses and permitted the fuel to heat up, which eventually caused an increase in radioactive releases. Finally, about ten days after the accident, liquid nitrogen was pumped into the space below the reactor vault. This not only cooled the fuel but, by preventing its further oxidation, essentially stopped the release of fission products.

While this essentially was the end of the accident, a major effort was subsequently required to decontaminate the environs of the reactor and the other reactor units. The damaged unit was eventually entombed in a large concrete structure separating it from the rest of the plant and from the environment.

Reactor units 1 and 2 at the Chernobyl complex have subsequently been restarted, while unit 3 is more heavily contaminated and remains shutdown.

5. Radiological Releases and Health Effects

5.1 Releases

The Chernobyl Unit 4 core contained a radioactive inventory of about 4×10^{19} becquerels (Bq) at the time of the accident. On the basis of radiation measurements and various technical analyses of samples taken from a 30 km radius around the Chernobyl plant and throughout the Soviet Union, the Soviets have estimated that about 1×10^{18} to 2×10^{18} Bq were released. This does not include contributions from the release of the noble gases xenon and krypton, which are believed to have been completely expelled. About 10 - 20% of the volatile radionuclides iodine, cesium and tellurium were expelled, and releases of the more refractory radionuclides, barium, strontium, plutonium, cerium, etc., amounted to 3 - 6%.

The release of radionuclides from the Chernobyl plant did not occur in a single event. Rather only 25% of the release (excluding noble gases) took place during the first day of the accident. The rest of the release of radioactive material occurred as a protracted process over a nine-day period.

5.2 Effects on the Workers

Two persons died immediately as a result of the explosion and fire. An additional 203 fire-fighting and power station personnel were diagnosed as suffering some degree of radiation sickness. Of these, 29 have subsequently died.

5.2.1 Radiation Burns

Most of the general literature on acute radiation effects has dealt with acute radiation syndrome, a condition which results from the destruction of blood-forming tissue in red bone marrow and of the lining of the gastrointestinal tract. However, a severe effect on many Chernobyl victims was radiation burns to the skin, caused by contact with radioactive dust from the damaged reactor core.

Altogether, 56 persons suffered from some degree of radiation burns. Of these, 20 cases were classified as "incompatible with life" (40% to 100% skin destruction) and 9 as "life-threatening". It appears that radiation burns were the main cause of death for many of the victims.

5.2.2 Acute Radiation Syndrome

203 persons suffered acute radiation syndrome, characterized by early vomiting and nausea, and later depression of blood cell and platelet counts. The approximate absorbed doses and the outcomes of the cases are summarized below.

Level of Severity	Dose (Grays)	Number Affected	Number Deceased
IV	6-16	22	21
III	4-6	23	7
II	2-4	53	1
I	1-2	105	0
Totals	-	203	29

Twenty patients with irreversible bone marrow damage were selected for transplants. Of these, seven died of burns before the transplants could be assessed. In another six, the transplants were rejected. The remaining patients showed some improvement, but it was not stable or permanent.

5.2.3 Internal Contamination

Although many victims absorbed radionuclides through inhalation, ingestion, or directly through damaged skin, the resulting internal contamination was not a major contributor to short term effects.

5.2.4 Neutron Exposure

The absence of the activation product sodium 24 in bioassay specimens indicated that the victims had not been exposed to neutron irradiation. The damage resulted primarily from gamma irradiation of the whole body and beta irradiation of the skin.

5.3 Effects on the General Population of the USSR

The location of the Chernobyl reactor site with respect to other countries and to the populated areas in its vicinity are shown in Figures 3 and 4 respectively.

5.3.1 Evacuation of Towns

Initially, residents of Pripyat and other nearby towns were ordered to stay indoors. Potassium iodide pills were distributed from door to door to reduce the uptake of radioiodines by the thyroid. However, by the evening of April 26, it was realized that a general evacuation would be necessary. Continuing emissions from the damaged reactor could lead to individual doses in excess of 0.25 sievert (Sv), the level for mandatory evacuation. By Sunday morning (April 27) radiation readings in the streets of Pripyat were rising to unacceptable levels. The evacuation began at 14:00, and by 16:45, all 49,000 residents of Pripyat had left. The town of Chernobyl (population 12,000) and the surrounding countryside, to a radius of 30 km around the reactor, were evacuated during the next two days. Many residents showed evidence of having ingested and inhaled radioiodine. Although there were enhanced levels of external exposure, there were no reported cases of acute radiation sickness.

Milk supplies were withheld whenever radioiodine levels surpassed 0.1 microcurie (3,700 Bq) per litre. Much of the 30 km area around the reactor still remains unfit for human habitation. Dose rates in some parts of this area are in excess of 0.2 milligray (mGy) per hour (normal background is 0.0001 mGy per hour).

Contamination of waterways does not appear to have been a serious problem, but the cooling pond of the damaged nuclear power station was contaminated.

5.3.2 Distribution of Iodine Pills

Distribution of pills to the public in the local area began at 09:00 on April 26. This timing, together with the amount administered, 250 milligrams per day, gave adequate thyroid blocking, because significant exposure to iodine did not commence until later that day, due to the meteorological conditions and the updraft from the reactor. Less than 3% of the affected population received thyroid doses in excess of 0.3 sievert (0.5 sievert per year to the thyroid is the ICRP recommendation for the maximum occupational exposure limit). However, it was not clear whether this was due to the distribution of iodine pills, or to the prompt evacuation and banning of milk supplies.

5.3.3 Long-Term Health Effects

There have been conflicting reports in the media on the number of eventual deaths due to cancer or genetic defects that may result from Chernobyl contamination. The following table summarizes the accumulated doses estimated by Soviet authorities for the evacuated population and for the European part of the USSR. On the basis of a risk coefficient of 1.25×10^{-2} fatalities per sievert, which is the approximate value obtained by extrapolating data on the consequences of high doses using a linear no-threshold dose-response hypothesis, the consequences listed below would be predicted. While it is possible that such an extrapolation would overestimate the consequences, it is apparent that the increase in the risk of contracting cancer to any individual is small, although when this small risk is integrated over a large population, a significant number of fatalities would be predicted.

	Evacuated Population	European USSR	
Type of exposure	External	External	Internal
No. of people	135,000	74.5 million	74.5 million
Average 50-year dose (mSv)	120	3.8	28
Collective dose (person-Sv)	16,000	290,000	2.1 million
Predicted fatal cancers	200	3,600	26,000
% of total predicted cancers	1.6	0.04	0.3

5.4 Effects Outside the USSR

The following is a brief summary of information made available at a meeting sponsored by the Committee on Radiation Protection and Public Health of the OECD, held in Paris on September 1-2, to discuss the impact of Chernobyl on member states.

	Canada	All OECD Countries (including Canada)
Total ^{131}I deposition (TBq)*	950	95,000
Total $^{134;137}Cs$ deposition (TBq)	420	16,000
Collective dose (person-Sv)	60	56,000
Possible cancer deaths	<1	700
% of total predicted cancers	<<1	<<1

* TBq = terabecquerel = 10^{12} becquerels.

Obviously the above table is incomplete, since it does not include eastern Europe or most of Asia.

Once again, it is apparent that the increased risk to any individual of contracting cancer is very small. More detailed information on the spread of radioactive material to Canada, and the radiological effects in Canada, is contained in reference 8.

6. Comparison of Chernobyl Design with CANDU Reactors

A detailed comparison of the design features of the Chernobyl Nuclear Power Power Plant and Canadian nuclear generating stations is given in Appendix B. Those aspects which are significant to reactor safety are further discussed below.

6.1 Plant Layout

The overall plant layout of Chernobyl is generally similar to that of the Pickering, Bruce and Darlington stations in Canada. All are multiple unit stations, and all have a common powerhouse running the entire length of the station which encloses the turbogenerators. The buildings which house the reactors are appended to the powerhouse.

6.2 Containment or Confinement

The confinement system at the Chernobyl station is described in section 2.4 and is of a different design from that of the CANDU reactor.

There are two basic containment designs of CANDU stations, one being a high pressure reactor building and the other an interconnected reactor building and vacuum building, in which the latter is designed to absorb the energy releases. Both designs contain dousing systems to condense the steam and further reduce the pressure. Both systems are required to withstand the peak pressure resulting from a break in the largest size of piping in the reactor's heat transport system.

Containments in CANDUs are superior in that essentially all of the heat transport system is in containment, but in neither design is the structure intended to withstand a reactivity accident with consequences as severe as those which occurred at Chernobyl.

6.3 Direct versus Indirect Cycle

The Chernobyl Nuclear Power Plant is a direct cycle reactor in that steam formed in the channels, after passing through the steam separators to remove the entrained water, goes directly to drive the turbogenerators. In the CANDU reactor design, a separate steam/feedwater system exists which extracts heat from the reactor primary heat transport circuit through large heat exchangers (boilers) and the steam produced in this secondary system is used to drive the turbines, thus being termed an indirect cycle reactor.

The direct cycle reactor is more sensitive to changes in both reactor and turbogenerator conditions. In the direct cycle, feedwater from the turbine condenser is returned to the steam separator, where mixing with the reactor coolant occurs. Changes in flow of either steam or water can therefore have a direct feedback on core inlet temperature and pressure, and hence reactivity separated in time only by a transport delay. This was an important design feature which contributed to the accident.

6.4 Moderator

Following the explosion at the Chernobyl reactor its graphite moderator became exposed to the air and began to burn. A great deal of effort was directed at extinguishing this fire, since its updraft was responsible for carrying large amounts of radioactive material into the atmosphere and surrounding areas.

The use of heavy water as a moderator eliminates both the possibility of a post-accident fire and the consequent wide-spread dispersal of radioactive material by this means if such an accident happened in a CANDU reactor.

6.5 Protective Shutdown Systems

Although the protective shutdown systems installed in a CANDU reactor are similar in their basic intent to the systems used in the RBMK reactor, they are very different in their implementation and overall philosophy.

Although the emergency rods used in the RBMK reactors fulfill a similar function to the CANDU safety systems by being used only for shutdown purposes, they are activated from the same detector measurements that are used for tripping the manual control rods. In addition, the effectiveness of the emergency shutdown system in Chernobyl depends on both the power at which the reactor is operated and on the shape of the neutron flux in the reactor. This contrasts with the Canadian licensing requirements where separate detectors must be used for the shutdown systems and where, to as great an extent as is practical, the tripping signals must be derived from both different types and methods of measurements. The shutdown systems in CANDU reactors are also required to be fully effective under all operating conditions.

In the RBMK reactors, as described earlier, a group of manual control rods is used to control the shape of the power distribution throughout the reactor but simultaneously fulfils a safety function by driving into the reactor to supplement the emergency rods if a dangerous condition is detected. In contrast, CANDU reactors are required to be equipped with shutdown systems which are independent from the control system.

The most important difference between the shutdown system designs is the speed at which the absorbing rods are inserted into the core. In the RBMK reactors, the rods are driven at 0.4 m/s into the reactor, so that in the case of the emergency rods it takes 20 s to insert them fully. A typical shutdown system in a CANDU reactor can shut the reactor down in less than 2 seconds.

6.6 Control Systems

The control systems in CANDU reactors are similar in principle to that used at Chernobyl, insofar as they both control in separate core regions or alternatively on an average or "bulk" basis.

Instead of inserting control rods to effect control, the CANDU automatic control system uses compartments partially filled with water. These are either filled or drained to change the neutron absorption in the different regions or zones of the core. In the bulk mode they drain or fill in unison.

An important difference is the response of each system when a controller approaches a physical limit at which its ability to control would be severely reduced. In the RBMK, if the controlling rods approach their limits, it is necessary to assist them manually to return to their operating range (by inserting or withdrawing manual control rods), while in a CANDU a separate group of control absorbers is automatically inserted when the zone control compartments are nearly full. Alternatively if the zone control compartments are approaching their drained limit, the CANDU control system automatically withdraws absorbers from the core until they come back in range. This major difference in design was an important factor in the loss of control which started the power runaway at Chernobyl.

6.7 Positive Void Reactivity

The positive void reactivity-feedback effect is a characteristic of the Chernobyl design which also exists in CANDU reactors (but not in most other types of reactor design). The implications of this are discussed in section 7.

The positive reactivity which would result from voiding the coolant from a single loop of a CANDU reactor is predicted to be approximately 7 mk, and in those reactors with two loops, would increase to approximately 11 mk if both were voided completely. This compares with a positive reactivity increase of 5 mk (estimated) for the voiding of both loops in an RBMK reactor operated at full power. However at the low power levels at which the Chernobyl accident occurred, and where the channels were almost filled with water, complete voiding would result in a very large reactivity increase of 30 mk. At similar power levels in a CANDU reactor, complete voiding would still result in a reactivity increase of 11 mk.

6.8 Dependence on Correct Operator Action

The design and operation of the Chernobyl reactor appears to have placed a high reliance on correct operator action. It appears that the operators were required to ensure on a continuous basis that sufficient manual control rods were inserted into the core to maintain a specified operating reactivity margin to ensure the effectiveness of the reactor shutdown system. Safe operation also depended on the avoidance of operation at relatively low power levels and high primary coolant flows at which the reactor was potentially very unstable.

Safe operation of all reactors depends on correct action by the plant operators. However, the CANDU design is more tolerant of operating errors than the Chernobyl design. In particular, automatic control systems are designed such that operator action is rarely, if ever, required to maintain their capability. In addition, the effectiveness of the shutdown systems in responding to a fast reactivity increase is not dependent on correct operator actions.

7. Implications of the Chernobyl Accident for CANDU Reactors

7.1 Positive Void Reactivity

A feature of the design of the Chernobyl reactor which played a very important part in the accident was the positive void coefficient of reactivity. As discussed in section 5.7, CANDU reactors also have a positive void coefficient of reactivity, although the effect is smaller (by about a factor of 3) than in the Chernobyl design when operating at low power levels. In the accident at Chernobyl, while the emergency shutdown system operated as designed, it was not designed to cope with the rapid increase in reactivity caused by the sudden increase in coolant void, and was therefore unable to act quickly enough to stop the increase in reactor power. If such an event, a rapid power increase with no effective shutdown system, were to occur in a CANDU reactor, the positive void coefficient could exacerbate the reactivity increase, and it is difficult to predict whether the reactor would shut itself down before power reached a level at which violent failure would occur. However, if a shutdown system operated as designed, it would terminate the reactivity increase and prevent serious consequences from occurring.

All the commercial reactors in the world have ways of adding significant reactivity, regardless of whether the void coefficient is positive or negative, and therefore require shutdown systems to cope with accidents in which reactivity increases. The design of all reactors and their licensing requirements must ensure that shutdown systems can act quickly enough, and add sufficient negative reactivity, to overcome the maximum reactivity which can be added in an accident.

The accident that occurred in the NRX reactor at Chalk River in 1952 was in some ways similar to the Chernobyl accident, although on a very much smaller scale. This accident was also caused by a loss of reactor control with which, in the circumstances, the automatic shutdown mechanisms were not able to cope. As a result of this accident, Canadian design standards and licensing requirements were developed to require reliable and fast-acting shutdown systems, capable of preventing accidents similar to those which occurred at NRX or Chernobyl with a high level of confidence.

The implications of the positive void coefficient of reactivity in CANDU reactors cannot therefore be considered in isolation, but must be examined together with the effectiveness of CANDU shutdown systems in coping with accidents in which the void reactivity is a factor. This is discussed in the following section.

7.2 AECB Requirements for Effectiveness of Shutdown Systems

Since the start of the commercial CANDU reactor program, it has been a requirement of the AECB that each reactor have a fast-acting shutdown system capable of shutting down the reactor quickly for a very wide range of possible accidents. Shutdown systems have been required to be independent of the reactor control systems, and to satisfy stringent requirements for reliability.

AECB Regulatory Policy Statement R-10 (reference 3), which applies to reactors licensed for construction after January 1, 1977, requires that there be two diverse and independent shutdown systems, each capable, on its own, of shutting the reactor down quickly when required. All the reactor units at Bruce G.S. "A", Bruce G.S. "B", Pickering G.S. "B", Darlington G.S., Gentilly 2 G.S., and Point Lepreau G.S. incorporate two such independent shutdown systems. AECB Consultative Document C-8 (reference 5), soon to be issued as a regulatory policy statement, confirms this requirement and specifies more detailed requirements for achieving effective and reliable shutdown systems.

The safety analyses for CANDU reactors are required to demonstrate the effectiveness of each shutdown system in shutting the reactor down sufficiently quickly to mitigate the consequences of a wide range of accidents from any power level. These postulated accidents include events such as rupture of a large pipe in the main reactor cooling system, partial or total loss of circulating pumps, and failure of the reactor control system, all of which have the potential to produce a rapid increase in the volume of steam in the reactor and a consequent positive reactivity effect due to the positive void coefficient. As a result of the above, the capability of the shutdown systems on CANDU reactors is required to be considerably greater than that of the emergency protective system at the Chernobyl reactor, particularly with respect to the speed at which they can shut down the reactor. Whereas the Chernobyl system was designed to insert negative reactivity at a rate of about 5 mk/s, a CANDU shutdown system can insert at least 24 mk in less than two seconds and for some designs the value is much higher.

Furthermore, the requirement that CANDU shutdown systems be fully independent from the reactor control systems means that their effectiveness must not depend on supplementary action by rods of the control system, as was the case at Chernobyl.

For those reactors which have two fully effective, reliable, and independent shutdown systems, the possibility that neither shutdown system would successfully shut down the reactor in an emergency situation is extremely remote. For this reason, the AECB accepts that complete failure to shut down in an accident does not need to be considered in the safety analyses of such reactors.

On the basis of the above, it is concluded that, for reactors which conform to current AECB licensing requirements, accidents similar to the one that occurred at Chernobyl are adequately protected against. The requirements for shutdown system design are adequate to ensure that a fast reactor shutdown will occur when required, even for those accidents in which the positive void coefficient of reactivity plays an important role.

Nevertheless, the severity of the accident at Chernobyl would suggest that it would be prudent to re-examine the safety analyses of CANDU reactors, with particular attention to events in which a rapid increase in the volume of steam may occur, or in which there may be a rapid increase in reactivity, to confirm that the shutdown systems are indeed sufficiently effective. In addition, it appears appropriate to examine possible configurations of reactivity devices to ensure that it is not possible to put the reactor in an unusual configuration in which the shutdown systems might be rendered less than adequately effective.

The conclusion for the four reactors at Pickering "A" is different, since these reactors each have only one shutdown system. While there are two shutdown mechanisms, dumping of the heavy water moderator and insertion of neutron-absorbing rods, these mechanisms are not independent of each other and rely on common instrumentation. In addition, the dumping of the moderator is relatively slow. At the time when Pickering "A" was originally licensed, there was not a requirement for two independent shutdown systems, but there was a requirement to analyze accidents in which a loss of reactor control or a loss of coolant was accompanied by complete failure of the shutdown system.

The original safety analysis for Pickering "A" included analysis of this postulated accident, and it was concluded that doses to the public would not exceed the AECB reference dose limits. However, a detailed quantitative analysis was not performed. It was largely because of the speculative nature of such analyses that the requirement for a second shutdown system was subsequently introduced on later reactors.

In an effort to produce a more sound basis for the safety of the Pickering "A" reactors, Ontario Hydro subsequently performed analysis of the effectiveness of the shutdown system with various levels of impairment, and concluded that the shutdown system could be effective in preventing serious consequences even if several shutoff rods failed to fall. These analyses improved the level of confidence that an accident with severe consequences would have a low probability of occurrence.

Furthermore, when the Pickering 'A' reactors were originally licensed for operation, the AECB requirement for the shutdown system was that the unreliability not exceed $10^{-2.5}$, i.e. failure to operate was to be no more frequent than one in about 300 attempts. The reliability of the shutdown system, as reported in Ontario Hydro's annual reports on station operation, has been considerably better than this. This gives further confidence that the risk from events involving failure to shut down is low.

Nevertheless, in view of the severity of the consequences of the Chernobyl accident, it would appear prudent to re-examine the safety of the Pickering "A" reactors, particularly with respect to accidents involving failure of the reactor control system or loss-of-coolant accompanied by unavailability of the shutdown system.

7.3 Compliance with Operating Procedures

The Chernobyl accident would not have happened if the reactor operators had avoided several serious violations of the rules for safe operation of the reactor. These violations are discussed in section 3 and listed in appendix C. The motivation for these violations was to complete the test successfully before the reactor was shut down for its maintenance outage. However, it is difficult to understand why the operators considered it appropriate to deviate so far from safe operating practices rather than to abort the test.

The Operating Licences of all power reactors in Canada include a condition that operation be in accordance with a document entitled "Operating Policies and Principles". This document defines the operating boundaries within which the plant may be operated safely, and specifies principles for ensuring that important safety-related equipment is maintained in a safe and operable state.

Both the policies of the plant owners, and the requirements of the Operating Licences issued by the AECB, forbid intentional violation of the "Operating Policies and Principles" both during normal operation and when conducting any special tests.

Within these safe operating boundaries, operating manuals define in detail the various procedures and lines of authority for operation of the plant. Station operators must be trained to operate the plant in accordance with these procedures, and senior operating staff are required to pass examinations set by the AECB to demonstrate their knowledge of the reactor and its operation.

The AECB maintains a staff of inspectors permanently at each major power reactor installation in Canada. A major function of these inspectors is to monitor the operation of the reactors, to ensure that operating procedures are being followed, and in particular to verify that operation is in accordance with the Operating Policies and Principles. Quality assurance audits of reactor operation are also performed periodically by head office staff of the AECB.

There have been occasions at CANDU reactor sites when operators have deviated from operating procedures and, in some cases, from the Operating Policies and Principles. When this has occurred, appropriate steps have been taken to prevent a recurrence. There have not been any occasions when operators have committed so many violations of such a serious nature as occurred at Chernobyl. It is the view of the AECB that the operator training and the disciplined approach to operation required both by the Canadian utilities and by the AECB would prevent violations of this nature in Canadian reactors.

The Chernobyl accident has underlined the importance of a disciplined approach to operation of nuclear reactors. In order to confirm that such an approach is followed in Canadian reactors, it is appropriate for both the AECB and the plant owners to continue to monitor and audit the performance of operators in complying with Operating Policies and Procedures, to review and if necessary, increase the frequency and extent of such compliance monitoring.

The AECB presently requires that the commissioning program for CANDU reactors be reviewed by the designers of the plant systems. Such reviews should be capable of detecting and correcting the shortcomings of a poorly-designed test, such as that which led to the Chernobyl accident. The Chernobyl accident showed the importance of such reviews, and of ensuring that approved test procedures are not modified without the modifications being properly reviewed.

It is appropriate to examine station procedures to ensure that they require the review and approval by appropriate personnel (including those with knowledge of the reactor design) before conducting tests on safety-related equipment or modifying the procedures for such tests.

7.4 Disabling of Safety Systems

Before the Chernobyl accident occurred, the operators disabled several important parts of the reactor's safety systems. These included the emergency core cooling system and parts of the emergency shutdown system. It appears that the design of the plant permitted the operators to do this quite easily.

Under certain circumstances (for example when a reactor is shutdown and depressurized), it is necessary to disable certain safety functions to prevent safety systems from operating unnecessarily. However, it is undesirable to enable operators to disable safety equipment without careful consideration of the implications, and approval of the action by station management. For this reason, AECB Consultative Documents C-7, C-8 and C-9 (references 4, 5, 6) stipulate that it shall not be readily possible for an operator to inhibit safety system action.

In the light of the Chernobyl accident, it would be prudent to re-examine the possible ways in which operators can disable special safety systems in CANDU reactors. These should then be reviewed to determine whether the design includes sufficient protection to prevent the operators from disabling the safety systems without first obtaining appropriate management and, if necessary, AECB approval.

7.5 Emergency Preparedness

The Chernobyl accident showed the importance of being adequately prepared for emergency situations, both on a reactor site and in the neighbouring vicinity off-site.

All power reactor licences issued by the AECB make reference to the on-site Radiation Emergency Procedures, and require that these be reviewed at least annually by the licensee, and that a report on this review be submitted to the AECB. The licensees should review their on-site emergency procedures in the light of the Chernobyl accident to determine if the need for any changes is indicated. In particular, the adequacy of monitoring instrumentation, both inside the plant buildings and in the immediate vicinity, should be considered.

With regard to off-site preparedness, the recommendations of the Emergency Planning Department of the Ontario Ministry of the Solicitor General are contained in reference 7. These recommendations should be considered by all those involved in emergency planning, both in Ontario and in other provinces where nuclear reactors are located in Canada. The AECB would participate in discussions with the appropriate provincial authorities to review cooperative arrangements in this area.

7.6 Fire-fighting in the Presence of Radiation

Most of the very high radiation exposures and resulting deaths caused by the Chernobyl accident were to people involved in fighting the fires which followed the initial explosions. The high doses received were mostly due to either direct gamma radiation or to contamination of clothing and skin with beta-emitting radioactive substances. This suggests that it is appropriate to review fire-fighting techniques in the presence of radiation, to determine how it would be possible or practicable to take measures to protect personnel from radiation exposures in such circumstances.

8. Conclusions

On the basis of this review of the accident at the Chernobyl nuclear reactor, AECB staff conclude that there is no important new information which would have a major effect on the licensing requirements for CANDU reactors as presently applied by the AECB. All important aspects of the accident and its causes have been considered by the AECB in the licensing process for currently licensed reactors.

However, in order to reinforce this conclusion, the designers and operators of Canadian nuclear reactors and the AECB should re-examine the pertinent safety aspects of CANDU reactors, and increase their vigilance in other areas. These are discussed in section 9 below.

This will serve to assist in ensuring that the risk posed by nuclear power reactors in Canada will continue to be acceptably low.

9. Recommendations

1) The safety analyses of CANDU reactors should be re-examined by the reactor designers and operators to confirm that shutdown systems are sufficiently effective under all possible conditions. Particular attention should be given to events in which a rapid increase in the volume of steam in the fuel channels may occur, or in which there may be a rapid increase in reactivity.

2) Various configurations of reactivity devices in CANDU reactors should be examined by the reactor designers and operators to ensure that it is not possible to put the reactor into a condition in which the shutdown systems might be rendered less than adequately effective. This should include an examination of the capability of the shutdown systems under conditions in which there are spatial variations in reactivity.

3) The safety of the Pickering 'A' reactors should be re-examined by Ontario Hydro and the AECB, particularly with respect to accidents involving failure of the reactor control system and loss-of-coolant accompanied by unavailability of the shutdown system.

4) Station operating procedures should be re-examined by plant owners to ensure that they require and receive sufficient review by all pertinent and responsible personnel before tests on safety-related equipment are carried out or procedures for such tests are modified.

5) The AECB and plant owners should review, and if necessary increase the frequency and extent of monitoring and auditing the performance of plant operators in complying with operating procedures, the Operating Policies and Principles, and the conditions of Operating Licences.

6) A study should be made by reactor designers and operators of the possible ways in which special safety systems can be disabled in CANDU reactors. These should then be reviewed to determine whether the design includes sufficient protection to prevent the disabling of safety systems without first obtaining appropriate management, and if necessary, AECB approval.

7) Plant owners should review their on-site emergency procedures in the light of the Chernobyl accident to determine if the need for any changes is indicated.

8) The recommendations of the Emergency Planning Department of the Ontario Ministry of the Solicitor General should be considered by all those involved in emergency planning both in Ontario and in other provinces where nuclear reactors are located.

9) A review should be carried out by plant owners of fire-fighting techniques to determine how it would be possible to take measures to protect personnel from any radiation hazard which may occur simultaneously with a reactor fire.

10. Acknowledgements

We acknowledge the assistance of Health and Welfare Canada, Environment Canada, the Ontario Ministry of the Solicitor General, and Atomic Energy of Canada Limited in reviewing and commenting on this report.

11. References

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Part I General Material
Part II Annexes 1 - 7

Annex 1 Water-Graphite Channel Reactors and Operating Experience with RBMK Reactors

Annex 2 Design of the Reactor Plant

Annex 3 Elimination of the Consequences of the Accident and Decontamination

Annex 4 Estimate of the Amount, Composition and Dynamics of the Discharge of Radioactive Substances from the Damaged Reactor

Annex 5 Atmospheric Transport and Radioactive Contamination of the Atmosphere and of the Ground

Annex 6 Expert Evaluation and Prediction of the Radioecological State of the Environment in the Area of the Radiation Plume from the Chernobyl Nuclear Power Station (Aquatic Ecosystems)

Annex 7 Data on Operational and Emergency Staff of the Nuclear Power Plant who were exposed to radiation: Size of the doses received and the consequences for health. Experience of treatment.

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Requirements for Shutdown Systems for CANDU Nuclear Power Plants, May, 1982

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Health Protection Branch, National Health and Welfare, December, 1986.

12. Glosssary

Radiation — energy in the form of invisible rays or particles which derive from nuclear transformations in materials. Some of these can, in passing through the body, damage normal cell structure by displacing their constituent atoms. This dislocation process is called ionization and the particular type of radiation which causes it is referred to as ionizing radiation.

Radiation Shield — a significant amount of radiation can be stopped by surrounding its source with an absorbing material. Such a shield can be made from different materials of which the most effective are usually heavy metals such as lead. Steel, concrete and water can also be used to stop different types of radiation.

Radiation Units — Activity - the strength of a radiation source is measured by the number of nuclear tranformations which occur each second.

One becquerel (Bq) is a transformation rate of one atom in each second.

This unit is usually too small to describe realistic radioactive source strengths and the following multiple units are more commonly used.

1 kilobecquerel - one thousand disintegrations per second
1 megabecquerel - one million disintegrations per second

Dose - the radiation derived energy which is absorbed in any substance which for example could comprise part or all of the human body.

A dose of one gray (Gy) corresponds to the absorption of one joule (1 watt-second) per kilogram of a substance.

Effectiveness - Observation of the effects of ionizing radiation on living cells shows that the magnitudes of specific biological effects per unit of absorbed dose vary with the nature of the radiation. In general a unit absorbed dose of neutrons or alpha particles produces significantly more damage than a unit absorbed dose of beta particles or X-or gamma rays.

This characteristic of a particular radiation type is referred to as its quality factor. For the purposes of specifying the effects of radiation the quantity dose equivalent is defined to take account of this dependence. Dose equivalent is the product of absorbed dose and quality factor.

Dose equivalent is measured in units of sievert (Sv).

Health Effects

Cancer Induction: The effect of low doses of radiation on people by direct observation has not been established. The effect, if there is one, is too small to be measured. Hence, it is necessary to extrapolate from the effects of high doses of radiation, where data are available, to low doses and it is assumed that this extrapolation is linear and has no threshold. Most experts think this assumption is conservative and that it overestimates the probable effects of low doses. However, by assuming no threshold, and by considering a very large population exposed to a small radiation dose, it is possible to calculate large numbers of possible casualties. The future casualties from Chernobyl have been estimated using these assumptions.

Thyroid Blocking

One of the major radiation hazards following a reactor accident in which its containment fails results from the significant quantities of radioactive iodine which may be released. These can concentrate in the thyroid gland giving rise to high radiation doses to that organ. To block this uptake of radioactive iodine, pills containing natural iodine can be administered immediately to the affected population after the accident.

Protection System: a system which monitors a selection of reactor and/or plant parameters and which initiates mitigating action in the event that one of them exceeds an abnormal level. When this happens the system is said to be tripped.

Containment: the boundary which surrounds a reactor's pressurized heat transport circuit and which, in the event of accidental fuel damage, prevents the release of radioactive materials to the surrounding environment. Containment is usually assisted by a means of suppressing pressure within its boundary.

Isotope: a material whose atoms are of the same species but which differ slightly in mass. There is for example one naturally occurring isotope of iodine and an additional 13 isotopes are created in nuclear reactors.

Turbogenerator: a machine in which steam is expanded through a turbine causing it to rotate. The rotational energy imparted in this way is used to drive an electrical generator.

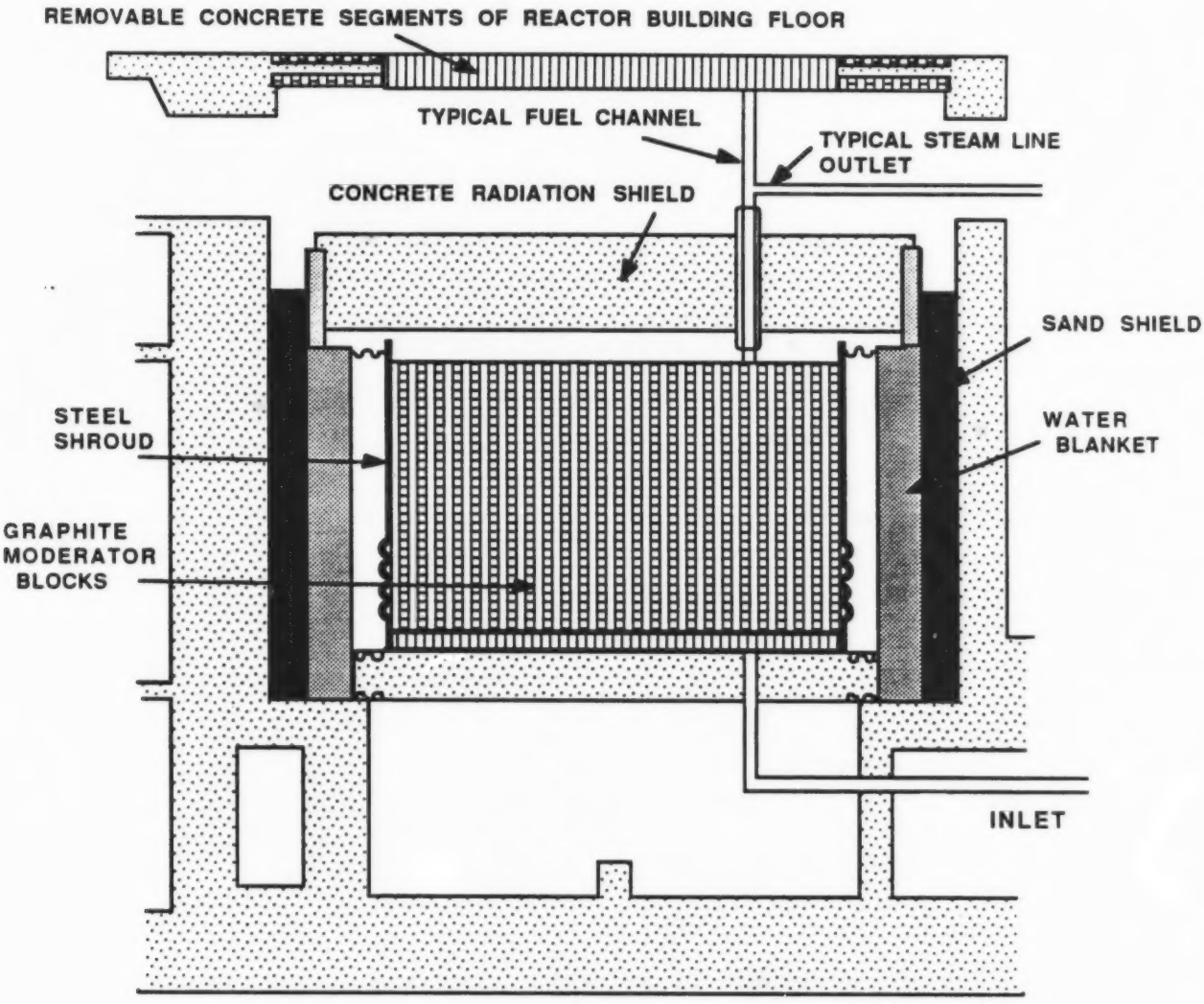

Fig. 1 CROSS SECTIONAL VIEW OF REACTOR VAULT

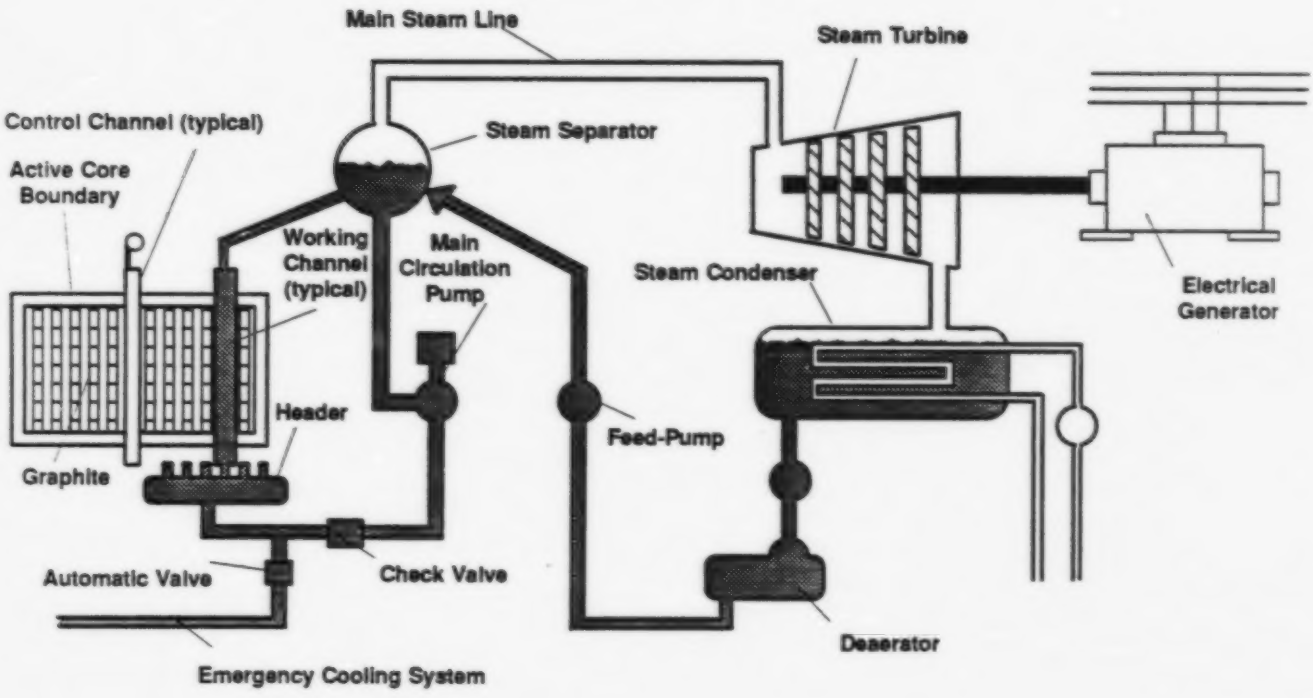

FIGURE 2 HEAT TRANSFER CIRCUIT

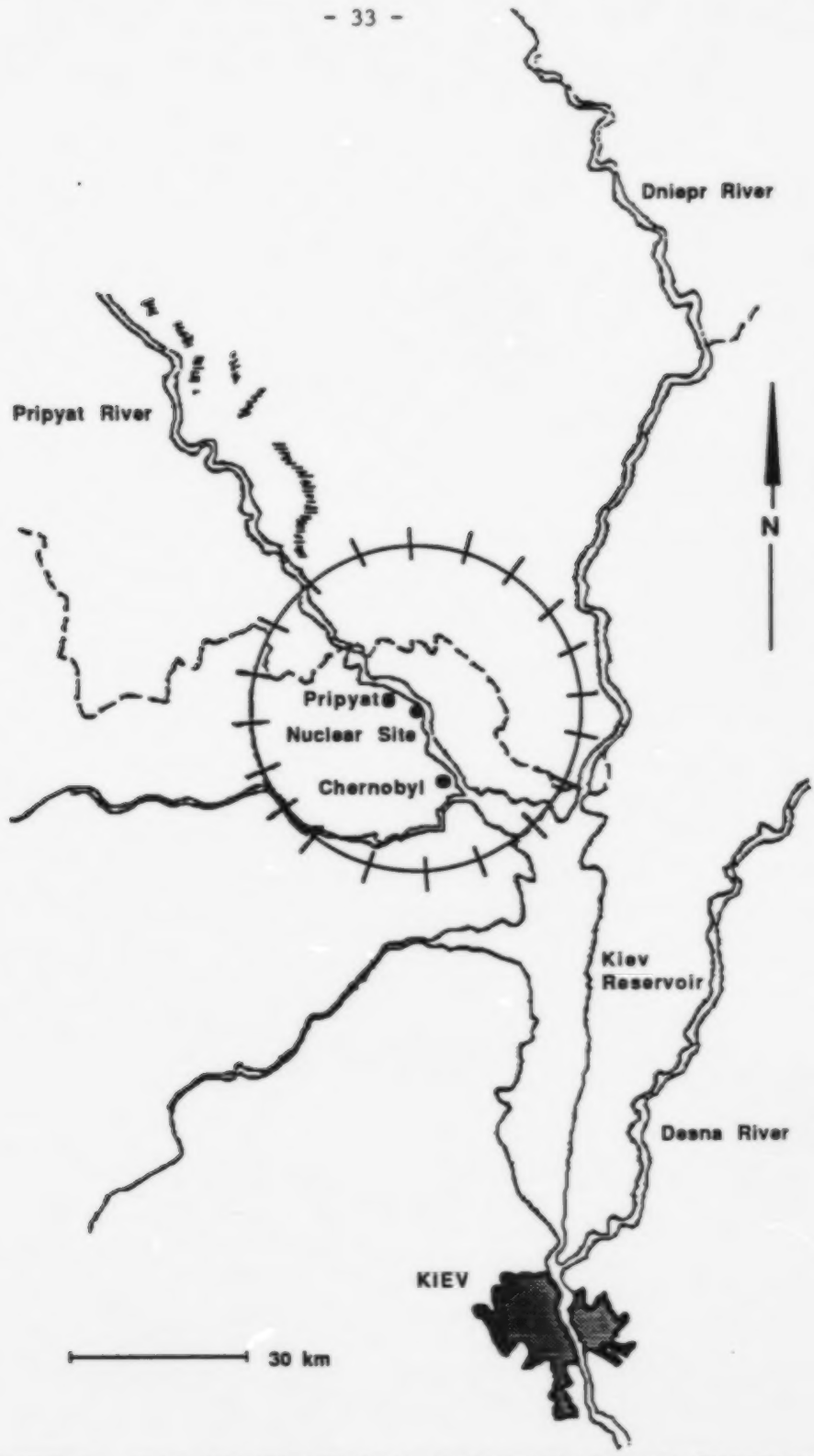

FIGURE 3 SITE OF CHERNOBYL NUCLEAR POWER STATION

FIGURE 4 COUNTRIES ADJACENT TO CHERNOBYL

APPENDIX A

TABLE I. ACCIDENT EVENT SEQUENCE

Time	Event	Interpretation	
		Result of event	Significance
25 April 1:00:00	The start of reactor power (a) reduction	Initial steps of test programme and planned maintenance outage	The slow reduction in power would help to reduce the effects of the buildup of xenon poison
13:05:00	Reactor power reduction stopped (b) at 50% of full power. Turbo-generator No. 7 switched off. Electric power requirements for Unit's needs switched to TG No. 8 (four main circulating pumps, two feed pumps, plus other equipment)		These components will run down with the TG during the test. Pump configuration at this time: four running from TG No. 8; two running from grid; two on standby connected to grid
14:00:00	The ECCS was isolated (c)	Done in accordance with the test plan because crew wishes to avoid spurious triggering	Safety principle violation, but blocked ECCS played no role in transient to point of core disruption. Might have been useful in the post-disruption period Note: Discussion by Soviet experts confirmed during the meeting that ECCS blocking was not necessary for the test
	Load dispatcher halted the (d) power reduction		The long hold in power would further reduce the buildup rate of xenon at test power level
	ECCS remained isolated (e)		

Time	Event	Interpretation	
		Result of event	Significance
23:10:00	Power decrease continued (f) towards the target level of 700-1000 MW(th)		In accordance with test procedure. This level was chosen to be above the minimum allowable operating power of the reactor (700 MW(th))
26 April 0:28:00	Operator error with transfer (g) from local (LAR) to global power (AR) control - hold power at required level not entered	Power reduced to 30 MW(th) owing to inability of the automatic control rods and lack of prompt operator action to compensate for the void due to power-flow mismatch	Negative reactivity added to system and more manual rods withdrawn to compensate
1:00:00	The reactor was stabilized at (h) a power of 200 MW(th)	Reactor operating below the minimum permissible power level Required reactivity reserve margin violated	No excess reactivity available to raise power
1:03	The fourth main cooling pump (i) powered from the grid was connected to the left loop of the heat transport system	Owing to low power and increased flow rate of coolant in the heat transport system, the coolant temperature approached saturation	This adds negative reactivity to the system, necessitating withdrawal of more rods to compensate

Time	Event	Interpretation	
		Result of event	Significance
1:07	The fourth main cooling pump (j) powered from the grid was connected to the right loop of the heat transport system	The flow rate in some of the main cooling pumps exceeded the permissible value There were significant deviations of the water level and steam pressure in the steam drums	Violation of flow vibration limits based on potential cavitation problems. Addition of both pumps caused further control rod withdrawal and further decrease in reactivity reserve margin
1:19:00	Operator increased feedwater (k) flow. About this time operator blocked the shutdown signals associated with steam drum level and pressure	Core subcooling increase results in more void collapse. Control difficulties throughout this period	
1:19:30	Onset of rise in water level in steam drum. The feedwater flow exceeded three times balanced value	The feedwater entering the system exceeds the steaming rate. The cooler water reached the core and reduced the steam quality and core void.	Steam drum level increases
	The automatic control rods went (l) up to the upper tie plate		Calculated core average void is now zero.
	The manual control rods were (m) raised		Addition of negative reactivity compensated by rod withdrawal
	The onset of SD steam pressure drop		

Time	Event	Interpretation	
		Result of event	Significance
1:19:58	The steam bypass valve was (n) closed	Slowdown in the rate of drop of the steam pressure	
1:21:50	The feedwater flow exceeded four times the balanced flow rate	Steam drum level still rising, pressure still falling	Control rod position constant as modelled. Reduction in pressure produces enough void to compensate additional feedwater flow
	Operator abruptly decreases the (o) feedwater flow		
1:22:10	Steam quality starts rising (p), automatic control rods start driving in, water level in steam drum stabilizes	Warmer water reaching core inlet produces a rise in average core void, control rods drive in to compensate	
1:22:30	Feedwater flow reduced to two-thirds of the balanced flow rate	Operator unable to stop feedwater flow rate at desired level owing to coarseness of control system, not designed for this operating regime	Control rods have moved in to compensate added reactivity of increased voiding
	The distribution of power (q) density and the positions of every control rod were printed out	This was done to establish the flux distribution and reactivity margin prior to beginning the test	Confirmation that the operational reactivity reserve margin was half of the minimum permissible, and the operator should have initiated immediate shutdown based on the computer printout

Time	Event	Interpretation	
		Result of event	Significance
1:22:45	Feedwater flow rate stabilized	Steam quality in the core stabilizing, pressure starts rising	
1:23:04	The personnel blocked the (r) two TGs-trip signal. Emergency stop valve to the (s) turbine was closed. The reactor continues operating at a power of 200 MW(th)	TG No. 8 test starts	Removal of last process safety system trip to allow test to be repeated. This trip would have saved the reactor. Operator aware he was inducing transient which required shutdown. (This was not provided for in the test programme)
1:23:10	One group of automatic control rods starts driving out	Core void decreasing because of increasing system pressure	
1:23:21	Two groups of automatic control rods begin reinsertion	Reduction of the coolant flow rate and the approach of the warmer water to the core	Both of these results lead to positive reactivity addition to the core. Control rods trying to balance this addition
1:23:31	Net reactivity increasing (t) with subsequent slow increase in reactor power	Control rods can no longer balance added reactivity	Power slowly rises; positive power coefficient accelerates reactivity imbalance

Time	Event	Interpretation	
		Result of event	Significance
1:23:40	Operator pushes AZ-5 button (reactor trip)	No apparent effect	
1:23:43	The triggering of the high power and short period alarms		The emergency protection is not efficient enough to prevent the reactor runaway
	The sharp growth of the calculated fuel temperature		Heat transfer crisis
1:23:44	Rapid increase in power		Calculated power reaches 100 times full power.

APPENDIX B

Comparison of Canadian Nuclear Generating Stations with the Chernobyl Nuclear Power Plant

ITEM	PICKERING GS A	BRUCE GS A	PICKERING GS B	BRUCE GS B	DARLINGTON GS A	600 MW Units	RBMK
Unit electrical output MWe (net)	515	740	516	750	881	600	1000
Number of turbines	4	4	4	4	4	1	8
Number of units	4	4	4	4	4	1	4
Direct or Indirect cycle	Indirect	Indirect	Indirect	Indirect	Indirect	Indirect	Direct
CORE AND FUEL DATA							
Channel attitude	Horizontal	Horizontal	Horizontal	Horizontal	Horizontal	Horizontal	Vertical
No. of channels	390	480	380	480	480	380	1661
Max. channel power MW(th)	5.12	6.5	5.37	6.5	6.4	6.5	3.25
No. of bundles/channel	12	13 (12 in Core)	12	13 (12 in core)	13 (12 in core)	12	1
Max. fuel element rating $\lambda\theta$.kW/m	4.2	4.2	4.25	4.2	4.2	4.2	1.69
Fuel enrichment	Natural	Natural	Natural	Natural	Natural	Natural	2% U_{235}
FUEL HANDLING	On-power	On-power	On-power	On-power	On-power	On-power	On-power
REACTOR	Pressure tube	Pressure tube	Pressure tube	Pressure tube	Pressure tube	Pressure tube	Pressure tube
Calandria vault	Air filled	Natural water filled steel shield tank	Natural water filled steel lined concrete	Natural water filled steel shield tank	Natural water filled steel shield tank	Natural water filled steel shield tank	Helium filled
End shielding	Steel plates and natural water	Steel balls and natural water	Steel balls and natural water	Steel balls and natural water	Steel balls and natural water	Steel balls and natural water	N/A
REACTIVITY MECHANISMS							
Adjuster rods (absorbers)	18 (cobalt)	-	21 (stainless steel)	24 (cobalt)	24 (stainless steel)	21	-
Booster rods (enriched uranium)	-	16	-	-	-	-	-
Control absorbers	-	4 (cadmium)	4 (cadmium)	4 (cadmium)	4 (cadmium)	(4 cadmium)	-
Vertical flux assemblies	8	20	14	27	27	26	
Horizontal flux assemblies	-	3	25	8	14	5	
REACTOR PHYSICS							
Void coefficient	1.14×10^{-4} Vol %	1.14×10^{-4} Vol %	1.14×10^{-4} Vol%	1.14×10^{-4} Vol%	1.14×10^{-4} Vol%	1.14×10^{-4} Vol%	2.0×10^{-6}% steam
Independent Critical Region	>2	>2	>2	>2	>2	>2	>10
Temperature coefficient	-4.2×10^{-6}/k	-4.2×10^{-6}/k	-4.2×10^{-6}/k	-4.2×10^{-6}/k	-4.2×10^{-6}/k	-4.2×10^{-6}/k	-1.2×10^{-5}/ C
Fast power coefficient of reactivity	-0.73×10^{-6}/MWt	-0.73×10^{-6}/MWt	-0.73×10^{-6}/MWt	-0.73×10^{-6}/MWt	-0.73×10^{-6}/MWt	-0.73×10^{-6}/MWt	-0.5×10^{-6}/MWt

APPENDIX B

Comparison of Canadian Nuclear Generating Stations with the Chernobyl Nuclear Power Plant - Con't

ITEM	PICKERING GS A	BRUCE GS A	PICKERING GS B	BRUCE GS B	DARLINGTON GS A	600 MW UNITS	RMBK
SAFETY SYSTEMS							
Primary shutdown mechanism	U162 21; U364 11, U162 50; U364 30	30 shutoff rods	28 shutoff rods	32 shutoff rods	32 shutoff rods	28 shutoff rods	24 emergency rods
Shutdown depth (mk)		39	65	61	61	65	
Shutdown rate (Time to Full Insertion) sec.	1.8	1.8	1.8	1.8	1.8	1.8	18.0
Secondary shutdown mechanism	Moderator dump	7 poison injection nozzles	6 poison injection nozzles	8 poison injection nozzles	8 poison injection nozzles	6 poison injection nozzles	No secondary shutdown
Dousing tank	In vacuum building	In vacuum building	In vacuum building	In vacuum building	In vacuum structure	In Reactor Building	Suppression pool
Emergency core cooling	From moderator system	Fast acting - high pressure (5.8 MPa) accumulators Long term - pumps	From high pressure pump system (Class IV power)	Fast acting - high pressure (5.8 MPa) accumulators Long term - pumps	From high pressure pump system (Class IV power)	Fast acting - high pressure (6.0 MPa) accumulators	Fast acting - high pressure (20.0 MPa) accumulators plus high pressure pump system Long term - pumps
Reactor building design pressure kPa(g)(psig)	41 (6)	69 (10)	41(6)	82.7 (12)	96.5 (14)	125 (18)	80
HEAT TRANSPORT SYSTEM							
Fluid	Heavy water	Heavy water	Heavy water	Heavy water	Heavy water	Heavy water	Light water
Inlet temperature °C (°F)	249 (480)	252/265 (483/509)	249.5 (480)	250/265 (482/509)	266	267 (511)	270°C
Outlet temperature °C (°F)	293 (560)	304 (579)	293.4 (560)	305 (581)	310	312 (594)	280.4°C
Quality at outlet header	5°C Subcooled	Subcooled	7°C subcooled	Subcooled	2.0%	4.0%	14.5%
Inlet pressure, core MPa(a) (psia)	9.55 (1385)	10.3 (1 492)	9.55 (1 386)	10.6 (1 542)	11.4	11.24	8.90
Heavy water inventory Mg/MW(e)	0.268	0.283	0.262	0.283	0.247	0.315	-
No. of steam generators	12	8	12	8	4	4	4 separators
Surface area per steam generator m^2 (ft^2)	1858 (20,000)	2368 (25,488)	1860 (20,000)	2400 (26,000)	4830	3127	Direct Cycle
No. of pumps	12 + 4 spare	4	12 + 4 spare	4	4	4	6 + 2 spare
Pump motor operating MW (hp)	1.23 (1 650)	6.41 (8 600)	1.17 (1 575)	6.41 (8 600)	6.70	4.78 (6407)	5.5
Pump motor rated MW (hp)	1.42 (1 900)	8.20 (11 000)	1.42 (1 900)	8.20 (11 000)	9.40	6.72 (9000)	4.3
Pressure Control	Feed and bleed	Pressurizer	Feed and bleed	Pressurizer	Pressurizer	Pressurizer	Steam bypass
Purification half life, min.	130	60	130	60	60	20	-
MODERATOR SYSTEM	Heavy water	Heavy water	Heavy water	Heavy water	Heavy water	Heavy water	Graphite
No. of Pumps	5 at 25%	2 at 100%	5 at 25%	2 at 100%	2 at 100%	2 at 100%	N/A
Heavy water inventory Mg/MW(e)	0.565	0.401	0.548	0.396	0.400	0.416	N/A
SECONDARY SYSTEM							
Steam flow Mg/s (10^6 lb/h)	0.81 (6.46)	1.17 (9.3)	0.81 (6.46)	1.17 (9.3)	1 315	.935 (7.42)	2.90
Steam pressure (drum) MPa(a) (psia)	4.09 (593)	4.38 (635)	4.09 (593)	4.38 (635)	5.07	4.7	6.46
Station efficiency (net) net output/total fission power	29.1%	29.1%	29.1%	29.1%	30.4%	29.0%	---

APPENDIX "C"

Errors and Violations of Procedures

Item

1. Operator requested by electrical grid controller to hold at 50% full power. At this point, the emergency core cooling system was isolated. The reactor remained at this power level, with the emergency core cooling system isolated, for the next 9 hours.

2. Consequences of carrying out test were not analyzed.

3. Operator erred with transfer from local to global power control causing reactor power to reduce further to 30 MW(t).

4. Reactivity reserve margin was exceeded and reactor power was below minimum allowable level.

5. Operator switched on two remaining main circulation pumps in order that the core would be continued cooled during test.

6. Operator blocked emergency protection on low separator water levels and steam pressure.

7. Operator blocked the reactor protection system which trips reactor when both turbogenerators are shutdown.